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A PRELIMINARY DESIGN STUDY OF A SMALL MOLTEN SALT REACTOR FOR EFFECTIVE USE OF THORIUM RESOURCE

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A preliminary design study has been made for a small molten salt power reactor, which use thorium resource effectively. A series of survey calculation are performed to obtain the over view of nuclear characteristics of the initial reactor core, widely changing the core design parameters. And a semi-optimized reactor is found out, taking into account the upper limit to the acceptable neutron exposure on graphite moderator and reactor vessel. Then, burn up characteristics are examined on this reactor operated with a simple fuel cycle at rated power of 150 MWe. The results suggest that the reactor can be operated with high performance for about 40 years, causing no severe troubles. And it produces little amount of plutonium and minor actinides which become long lived radioactive waste.

Keywords Molten salt reactor, Nuclear characteristics, Burn up, Thorium, Uranium-233

1 INTRODUCTION

Light water reactors(LWRs) are well established and economical systems which supply more than 19% of total electric energy consumed in the whole world. They will continue to be one of the main systems for power generation in the next century. However, there are concerns about energy resource, reactor safety, social and environmental problems, such as the large amount of plutonium(Pu) and the long lived radioactive waste accumulated from LWRs operated worldwide. Some of these issues will be resolved by adoption of the reactor operated with thorium(Th) uranium(U) fuel cycle.

Molten salt reactor fueled with U-233 is one of the best reactors which are operated with Th-U fuel cycle. This kind of reactor had been researched and developed extensively at Oak Ridge national laboratory(ORNL) from 1950's to 1970's. They built and operated a 10 MWth experimental reactor in 1965, and then performed a conceptual design study of the molten salt breeder reactor (MSBR) (Rothental, 1970). The reactor was prominent in performances, that is, fuel breeding ratio was 1.06 and fuel doubling time was about 22 years. To obtain high breeding ratio, MSBR adopted continuous fuel reprocessing by installing complicated facilities, which might take place some difficulties in development and economical operation of the reactor.

In this paper, the authors have performed a preliminary design study to examine the conceptual feasibility of a small molten salt power reactor for effective use of Th resource. It has 150 MWe output capacity. Though the core structure in the present study resembles the one reported previously (Mitachi, 1990), the authors intended to make the reactor core structure simpler, to operate it with minimum fuel reprocessing and minimum graphite moderator replacement, in order to improve economical efficiency in reactor construction and reactor operation. Critical characteristics of the reactor are made clear at the beginning of the reactor run by using the latest analysis codes and nuclear cross section data. Then, burn up characteristics of the reactor are estimated for 14000 days reactor run at rated power, assuming the reactor operation with a simple fuel cycle mode, namely, fuel salt is reprocessed once in every 6 years by batch process, and graphite moderators are replaced in every 15 years.

2 CHARACTERISTICS OF THE INITIAL REACTOR CORE

2.1 Core configuration

A schematic diagram of the reactor core is depicted in Fig.1. It is a graphite moderated thermal reactor which is simply composed of core and reflector surrounded by neutron absorber made of boron-carbide. The reactor vessel is made of modified Hastelloy N. The fuel salt containing fissile and fertile materials flows slowly through passages built in the graphite elements in every zones i.e. the core, the reflector and the absorber. The moderators and reflectors are made of high density graphite of which density is 1.84 g/cm^3. The graphite volume fraction is changed in each zone, adjusting fission and capture reaction rates. The fuel salt is a molten salt mixture, mainly composed of the fluorides of lithium (Li), beryllium (Be), Th and U. Its composition in the initial reactor core is indicated in Table 1. The reactor can be operated for a long period without continuous fuel reprocessing and without graphite moderator replacement.

2.2 Calculation method

The nuclear characteristics of the initial core, such as the effective neutron multiplication factor (k_{eff}), neutron flux and fission rate distributions in the core, are calculated by using the standard reactor analysis code (SRAC95) which was developed at Japan atomic energy research institute (JAERI)(Okumura, 1996). The SRAC95 analysis is performed with 30 group nuclear cross section data obtained from nuclear data file JENDL3.2 which is prepared at JAERI.

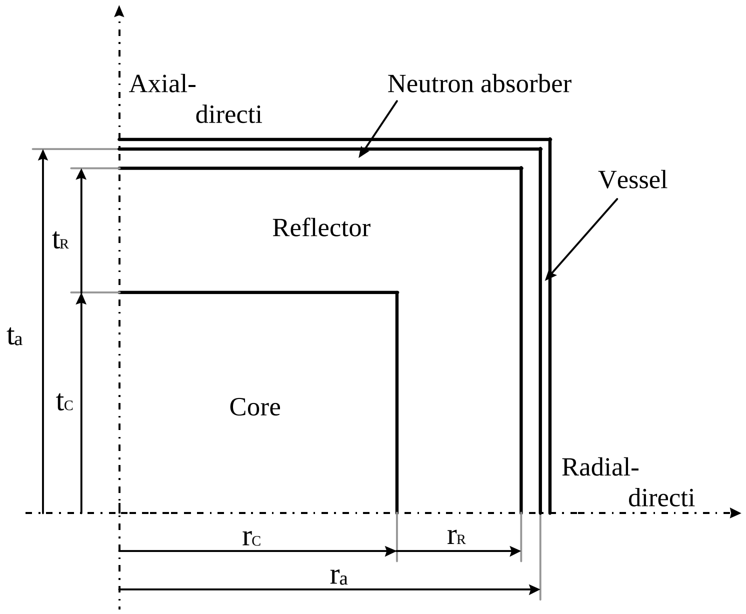

Figure 1 *Reactor core configuration.*

Table 1 *Fuel salt composition at the beginning and the end of reactor run.*

Burn Up (days)	Li (mol%)	Be (mol%)	Th (mol%)	U (mol%)	Other elements (mol%)
0	71.78	16.00	12.00	0.22	0.00
14000	71.40	15.87	11.87	0.36	0.50

2.3 Semi-optimized reactor core

It is desirable for the reactor to have higher fuel conversion ratio, smaller reactor volume, smaller U-233 inventory and lower neutron radiation damage. A series of survey calculations are first conducted to obtain the over view of the nuclear characteristics of the initial reactor cores, widely changing the parameters such as U-233 concentration, the fuel salt composition, the graphite volume fraction, the core geometry, etc. As a result, a semi-optimized reactor core, which we gave a nickname Fuji, is designed for the condition of acceptable exposure on graphite, namely, $6.3\times10^{13} n/cm^2/s$ for the fast neutron flux higher than 50 keV. The exposure on modified Hastelloy N is limited to $1.2\times10^{11} n/cm^2/s$ for the fast neutron flux higher than 0.8 MeV, and $6.0\times10^{12} n/cm^2/s$ for the thermal neutron flux lower than 0.18 eV. These limitations on the irradiation cited from the reference (Rothental, 1970) can ensure the reactor operation for about 15 years without graphite moderator replacement. The reactor vessel can be used for about 30 years under these neutron irradiation.

Temperature coefficient of reactivity must be negative in order to realize the safe and stable reactor operation. The previous reactor core (Mitachi, 1990) might have a problem in this view point. In the present study, graphite volume fraction in the core is decreased, and U-233 concentration in the fuel salt is increased so as to make the temperature coefficient of reactivity more negative.

Several important parameters of Fuji are listed in Table 2. The reactor vessel is 5.3m in diameter and 5.5m in height. Inventories of U-233, Th and graphite are 0.54 ton, 29.2 ton and 182 ton, respectively. The reactor can generate 350MW_{th} of heat. The temperature of fuel salt at the outlet of the reactor becomes 980K, capable of generating super heated steam with 25 MPa pressure and the temperature of 810K. The output of electricity becomes 150 MW_e by installing the steam power generating system with net thermal efficiency of 43%.

Table 2 *Principal design parameters of the reactor core (Fuji).*

Thermal capacity	350 MW_{th}	Power density	7.0 kW_{th}/liter
Net electric generation	150 MW_e	Neutron multiplication factor	1.024
Thermal Efficiency	43%		
Reactor vessel		Fuel salt	
Diameter/Height	2.65/2.75 m	$^{233}UF_{3.91}$ concentration	0.22 mol%
Core		ThF_4 concentration	12.0 mol%
Maximum radius	2.0/2.0 m	BeF_2 concentration	16.0 mol%
Graphite fraction	70 vol%	Volume in reactor	15.7 m^3
Reflector		Total volume	20.2 m^3
Thickness Rad./Axi.	0.6/0.7 m	Flow rate	0.55 m^3/s
Graphite fraction	99 vol%	Temperature In/Out	840/980 K
Maximum neutron flux		Inventory	
Graphite (>52kev)	6.08×10^{13} n/cm^{2}s	U	0.54 ton
Metal (>0.8Mev)	0.58×10^{11} n/cm^{2}s	Th	29.2 ton
(<0.18ev)	0.47×10^{12} n/cm^{2}s	Graphite	182 ton

3 BURN UP CHARACTERISTICS

3.1 Calculation method

The burn up characteristics of the reactor are calculated using the same procedure reported previously (Mitachi, 1995). In the molten salt reactor, the fuel salt is kept homogeneous by its circulation through the core, pumps and heat exchangers. But the fuel salt composition and the fissile concentration always change, because of the nuclear reaction in the core and the additional fuel supply from outside of the reactor. In order to simulate this continuous change of the composition and the concentration in the fuel salt, the burn up time is divided into a series of short time intervals. The fission rates in every zones in the core are first calculated by SRAC95 at the beginning of the time interval. Then, the irradiation and decay calculations are carried out for every time intervals, by using ORIGEN2 which was developed at ORNL for calculating the build up, decay and processing of radioactive materials(Croff, 1980). The 1 group nuclear cross section data, which are necessary in the depletion calculation by ORIGEN2, are obtained from JENDL3.2. These calculation procedures for one time interval are repeated, covering all reactor run time.

3.2 Reactor operation for 2000 days

With the reactor operation, fissile materials decrease, and fission products accumulate in the fuel salt, causing lower reactivity of the core. Hence, the fissile materials should be fed in order to keep the steady reactor operation. In the present study, fissile material is U-233, which is supposed to be fed into the fuel salt in the form of frozen eutectic salt i.e. $LiF(73mol\%)-UF_4(27mol\%)$. This composition of the lithium-uranium eutectic salt might be suitable to roughly compensate for the change of Li and fluorine (F) caused by burning including fission, taking place only a little change on fuel salt composition. In the calculation, the eutectic salt is assumed to be fed into the fuel salt periodically, so as to keep the excess reactivity of the core in the range from 1.5% to 2.0% all the time, as shown in Fig.2. The feeding cycle is about once in 10 days in the beginning of the reactor run, and once in 20 days after around 100 days. The feeding rate of uranium-233 becomes 500 g/day in the beginning, and decrease to about 90 g/day after around 200 days.

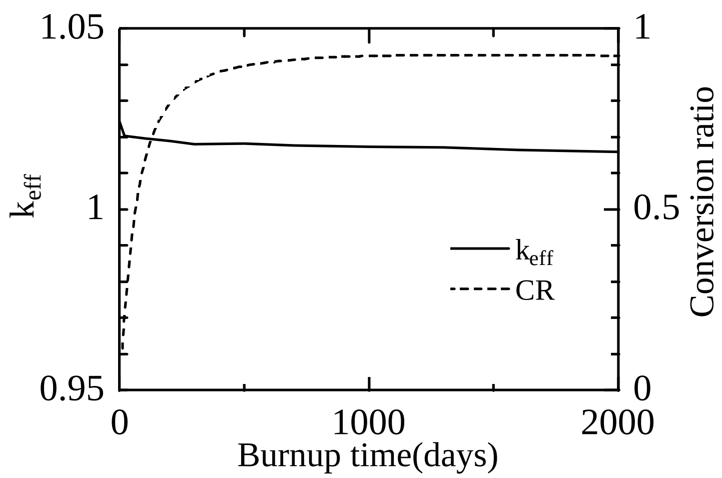

Figure 2 *Changes of neutron multiplication factor(k_{eff}) and fuel conversion ratio(CR) during 2000 days reactor run.*

We define fuel conversion ratio (CR) as the ratio of produced amount of fissile isotopes to consumed amount of fissile isotopes during the reactor operation. As shown with the broken curve in Fig.2, CR is very low at the beginning of the reactor run, and increase to about 0.9 at around 500 days. This is because the production of U-233 in the reactor is delayed owing to protactinium (Pa) decay time of about 27 days. The CR becomes 0.93 at 2000 days.

With the burn up of fuel salt, component materials of fuel salt are consumed, taking place the change of fuel salt composition. Because the change of the composition might cause chemical attacks on reactor vessel and graphite moderator, so that Li, Th, F and Be should be added periodically in order to adjust the fuel salt composition. In the present calculation, Li, Th and F are supposed to be added as LiF, $LiF(71mol\%)-ThF_4(29mol\%)$, and F_2 gas, respectively. Be is added as BeF_2 or Be, depending on lack or surplus of F in the fuel salt. The component materials have so large inventories in the reactor that we can adopt a long adding cycle to meet requirement for a small change in the fuel salt composition. The adding cycle is once in 100 days in the beginning of reactor run, and once in 300 days after around 700 days. Also, with the same cycle, the nuclear cross sections are renewed in ORIGEN2 analysis.

3.3 Fuel salt reprocessing

Long term reactor operation increases fission products accumulated in the fuel salt, that might cause some undesirable effects, such as lower neutron economy, precipitation trouble of fission and capture products, etc. To prevent these undesirable effects, the fuel salt should be reprocessed periodically, for example once in several years. In the present study, the reprocessing is assumed to be performed once in every 2000 days. In this reprocessing, all products produced by fission and capture reactions, except uranium isotopes, are removed out of the fuel salt. Also, the composition and the volume of fuel salt are reset to ones of initial reactor.

3.4 Reactor operation until 14000 days

Burn up calculation for 2000 days succeeded by fuel salt reprocessing is repeated until 14000 days reactor run at rated power of 150MWe. In Table 1, the fuel salt composition at 14000 days is compared to the one at the beginning of the reactor run. The composition of fuel salt is changed little by 14000 days reactor operation, owing to the fuel feeding in the form of the eutectic salt, the periodical adding of the fuel salt components and the fuel salt reprocessing in every 2000 days.

The changes of reactor characteristics are listed in Table 3. These are the characteristics just before the fuel salt reprocessing. The neutron multiplication factor k_{eff} is in the range from 1.015 to 1.020, except in the beginning of the reactor run. The fuel conversion ratio CR averaged in 14000 days are around 0.92, indicating that the reactor is able to use neutrons very efficiently. The temperature coefficient of reactivity α_T changes from -1.5×10^{-5} to -1.8×10^{-5}

1/T, confirming safe and easy reactor control.

Concerning to irradiation damage, fast neutron flux higher than 52 keV is less than $6.1\times10^{13} n/cm/s$ all the time in the core. Fast neutron flux higher than 0.8 MeV is less than $0.75\times10^{11} n/cm/s$ and thermal neutron flux lower than 0.18 eV is less than $0.47\times10^{12} n/cm/s$ on the inner surface of reactor vessel. These results are less than the design criteria ensuring that the graphite moderator can be used without replacement for about 15 years and the reactor vessel can be used for about 45 years.

In Table 4 are listed amounts of initial inventories of heavy metal, U-233 and Th fed during 14000 days, and final remains at 14000 days. The last item means total amounts of materials obtained through the fuel salt reprocessing minus total amounts of materials needed for start up of next 2000 days reactor run. Fuji consumes 338 kg of U-233, and produces 3 kg of Pu, 2.1g of Am and Cm in 14000 days operation at the rated power of 150 MWe.

Table 3 *Changes of reactor core characteristics during 14000 days run.* α_T *is temperature coefficient of reactivity.* ϕ_G *denotes the maximum neutron flux in the reactor core, and* ϕ_M *on the inner surface of the reactor vessel.*

Burn up (days)	K_{eff}	CR	α_T (1/K) [$\times10^{-5}$]	ϕ_G ($n/cm^2/s$) >52keV [$\times10^{13}$]	ϕ_M ($n/cm^2/s$) <.18eV [$\times10^{12}$]	ϕ_M ($n/cm^2/s$) >.8MeV [$\times10^{11}$]
0	1.024	0	-1.8	6.1	0.47	0.58
2000	1.016	0.93	-1.5	5.9	0.47	0.70
4000	1.016	0.93	-1.5	5.8	0.47	0.72
6000	1.017	0.93	-1.5	5.8	0.47	0.73
8000	1.017	0.92	-1.5	5.8	0.47	0.74
10000	1.016	0.92	-1.6	5.8	0.47	0.74
12000	1.017	0.92	-1.6	5.8	0.47	0.75
14000	1.017	0.92	-1.6	5.8	0.47	0.75

Table 4 *Initial inventories, net feeds and final remains of actinides for 14000 days run.*

	Th (ton)	U_{fis}+Pa_{233} (ton)	Pu (kg)	Am (g)	Cm (g)
Initial inventories	29.2	0.539			
Total net feed	5.6	0.539			
Final remain	29.7	0.740	2.97	1.9	0.16

The 14000 days reactor run resulted fission products generation in the fuel salt as shown in Table 5. They are classified into fission gas, dissolvable fission products and non-dissolvable fission products. Fission gas can be easily isolated directly from the fuel salt by helium gas bubbling, and tritium from cover gas of secondary system mostly.

Dissolvable components except three valence salts are about 0.14 mol% and sufficiently low concentration in the fuel salt, so that they would not take place any chemical problems. Considering the solubility limit, the total amount of dissolvable three valence salts should be less than 1 mol% in the fuel salt. As one can see in the table, Pu is very low and the total of three valence elements is about 0.16 mol% even at 14000 days, occurring no precipitation of the components.

The total amount of non-dissolvable components is presumed 110 kg. Some part will be removed by filtration, and residual deposit will not disturb the salt flow, because the salt volume fraction is fairly high as 30 vol% in the core. The general behavior of fission products will not tentatively have any serious problems in the reactor operation.

Table 5 *Amounts of fission products included in the fuel salt at 14000days.*

(1) Gaseous species:			(2) Undissolvable components:			
	Xe	0.25 g	Mo	63 kg	Rh	4 kg
	Kr	0.07 g	Tc	4 kg	Pd	5 kg
	T	149 g*	Ru	34 kg	Ag	0.2 kg
					sub total	110 kg
(3) Dissolvable components:						
a) Stable salts except three valence ones:						
	I	8.9 kg	0.0060 mol%			
	Br	0.8 kg	0.0009 mol%			
	Cs	13.0 kg	0.0088 mol%			
	Rb	0.5 kg	0.0005 mol%			
	Sr	16.6 kg	0.0172 mol%			
	Ba	10.9 kg	0.0073 mol%			
	Zr	95.0 kg	0.0946 mol%			
					sub total	0.1353 mol%
b) Three valence salts:						
	Pa	13.8 kg	0.0055 mol%			
	U^{+3}	20.5 kg	0.0081 mol%			
	La	24.6 kg	0.0164 mol%			
	Ce	61.3 kg	0.0402 mol%			
	Pr	27.9 kg	0.0183 mol%			
	Nd	78.7 kg	0.0504 mol%			
	Pm	2.4 kg	0.0015 mol%			
	Sm	10.4 kg	0.0065 mol%			
	Y	9.7 kg	0.0100 mol%			
	Pu	0.8 kg	0.0003 mol%			
					sub total	0.1572 mol%

**This numerical value is total amount of tritium produced during 14000 days run.*

4 COMPARISON TO BOILING WATER REACTORS

In Table 6, molten salt reactor is compared with Boiling Water Reactors (BWRs) in view of the fissile consumption and radioactive waste production. Using uranium oxide fuel which is burned until 45 GWd/ton, the boiling water reactor BWR(U) consumes 801 kg of uranium-235, and produces 230 kg of Pu, annually for 1 GWe of electricity generation. The total production rate of Am and Cm becomes 25 kg/GWe/year (Hida, 1992). If charged with mixed oxide fuel in place of uranium oxide fuel, the reactor BWR(MOX) is able to burn 649 kg of Pu, but the reactor produces 133 kg of Am and Cm, annually for 1 GWe generation (Hida, 1992).

Fuji consumes 58.7 kg of U-233, and produces 0.52 kg of Pu, annually for 1 GWe generation. Also, it produces Am and Cm by 0.36g. Net fissile consumption rate is only about 59 kg/GWe/year for Fuji, but the net rates are 659 kg/GWe/year for BWR(U), and 639 kg/GWe/year for BWR(MOX). Fuji has higher fuel conversion ratio and produces much U-233 in its core, so that it needs little amount of fissile supply in the reactor operation. The production rates of Am and Cm for 1 GWe and 1 year are 0.36g for Fuji, 25 kg for BWR(U) and 133 kg for BWR(MOX), respectively. By adopting Th-U fuel cycle, Fuji produces little amount of actinides, which become long lived radioactive waste, compared to BWRs used today.

Table 6 *Production rate of uranium, plutonium and minor actinides. BWR(U) denotes the boiling water reactor operated with uranium oxide fuel, and BWR(MOX) the reactor operated with mixed oxide fuel. Fuji is the molten salt reactor studied in this paper.*

	BWR(U)	BWR(MOX)	FUJI
Electricity generation	1.0 GW_e	1.0 GW_e	0.15 GW_e
Thermal output	3.0 GW_{th}	3.0 GW_{th}	0.35 GW_{th}
Net production rate per 1 Gwe year			
U-fissile	-801 kg	-22 kg	-58.7 kg
Pu-fissile	142 kg	-617 kg	0.04 kg
total Pu	230 kg	-649 kg	0.52 kg
Am+Cm	25 kg	133 kg	0.36 g

5 CONCLUDING REMARKS

A preliminary design study has been made for a small molten salt power reactor named Fuji, which is operated with a simple thorium uranium fuel cycle. The nuclear characteristics of the reactor have been analyzed theoretically, and the following facts have been revealed.

1.The reactor can be run for about 40 years at rated power, utilizing thorium resource effectively.

2.Fuel conversion ratio is about 0.92, and we can run the reactor with little fuel feeding.

3.The temperature coefficient of reactivity becomes less than $-1.5 \times 10^{-5} / T$.

4.Dissolvable fission products have sufficiently low concentration in the fuel salt, even after 14000 days reactor run, causing no chemical problems and no precipitation troubles.

5.The reactor produces little amount of actinides, which becomes long lived radioactive waste, compared to light water reactor used today.

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